U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.1.1 - 15.1.4 DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (~~RSB~~SRXB[1])

Secondary - ~~None~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]

I. AREAS OF REVIEW

A number of ~~transients~~ events[3] which are expected to occur with moderate frequency, and which involve an unplanned increase in heat removal by the secondary system, are covered by this Standard Review Plan (SRP)[4] section. Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. The power level increase will lead to a reactor trip. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure.

Each of the ~~transients~~ initiating events covered by this SRP section should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The ~~transients~~ initiating events to be evaluated include:

1. Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs)

   a. Feedwater system malfunctions that result in a decrease in feedwater temperature.

b. Feedwater system malfunctions that result in an increase in feedwater flow.

c. Steam pressure regulator malfunctions or failures that result in increased steam flow.

2. PWRs Only

a. Inadvertent opening of a steam generator relief or safety valve.

The topics covered in the primary review include: postulated initial core and reactor conditions which are pertinent to feedwater system malfunctions, pressure regulator or pressure relief valve malfunctions, methods of thermal and hydraulic analysis, postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor system components, functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the transient analysis are reviewed to ensure that the values of pertinent system parameters are within the ranges expected for the type and class of reactor under review. The parameters include: core flow and flow distribution, channel heat flux (average and hot), minimum critical power ratio (MCPR), departure from nucleate boiling ratio (DNBR), vessel water level, thermal power, vessel pressure, steam line pressure (for BWRs), steam line flow (for BWRs), feedwater flow (for BWRs), and reactivity.

The sequence of events described in the SAR for these transients is reviewed by ~~RSB~~SRXB.[5] The ~~RSB~~SRXB[6] reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The analytical methods are reviewed by RSB to ascertain whether mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the ~~RSB~~SRXB[7] reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

Review Interfaces[8]

The ~~RSB~~SRXB[9] will coordinate other branch evaluations that interface with the overall review of the transient analyses as follows:

A.[10] The Instrumentation and Controls ~~Systems~~ Branch (~~ICSB~~HICB[11]) reviews the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems as part of its primary review responsibility for SRP Sections 7.2 through 7.5. For B&W plants, HICB also reviews the applicant's design criterion for the allowable number of actuation cycles of the emergency core cooling system and the reactor protection system consistent with the expected occurrence rates of severe overcooling events, considering both anticipated transients and accidents (Reference 24).[12]

B. The ~~Core Performance Branch (CPB) upon request from RSB~~SRXB~~,~~[13] reviews the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage limits; and reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4.

C. The ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[14] using fuel damage results provided by ~~RSB~~SRXB[15] evaluates the radiological consequences associated with the fuel failure.

D. The review of the Technical Specifications is coordinated and performed by the ~~Licensing Guidance Branch (LGB)~~Technical Specifications Branch (TSB)[16] as part of its primary review responsibility for SRP Section 16.0.

For those areas of review identified above ~~as being reviewed~~[17] as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary[18] review branch.

II. ACCEPTANCE CRITERIA

The ~~RSB~~SRXB[19] acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10 (GDC 10),[20] as it relates to the reactor coolant system being designed with appropriate margin to ~~assure~~ ensure[21] that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15 (GDC 15),[22] as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ~~assure~~ ensure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences.

C. General Design Criterion 20 (GDC 20), as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences.[23]

~~C~~D.[24] General Design Criterion 26 (GDC 26),[25] as it relates to the reliable control of reactivity changes to ~~assure~~ ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ~~assuring~~ ensuring that appropriate margin for malfunctions such as stuck rods are accounted for.

~~D. TMI Action Plan items II.E.5.1 and II.E.5.2 of NUREG-0718 as they relate to assuring that any design modifications that result from the resolution of these Action Plan items are properly accounted for in the analyses.~~[26]

The basic objectives of the review of the transients which result from an increase in heat removal are:

1. To identify which of the moderate-frequency* ~~transients~~ events that result in increased heat removal are the most limiting.

2. To verify that, for the most limiting ~~transients~~ events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the requirements of ~~GDC~~ General Design Criteria[27] 10, 15, 20,[28] and 26 for incidents of moderate frequency are:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 2).

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. An incident of moderate frequency in combination with any single active component failure, or single operator error,[29] shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2) that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

5. To meet the requirements of General Design Criteria 10, 15, 20,[30] and 26 the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

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*The term "moderate-frequency" is used in this SRP section in the same sense as in the descriptions of design and plant process conditions in References 9 and 10.

6. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 ~~(Ref. 12)~~.[31]

The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 5 through 8 are acceptable. BWR and ABWR pressurization events should be evaluated using the models provided in References 28 and 17, respectively. Reference 18 provides acceptable models for analysis of other (non-pressurization) transients for the ABWR.[32] References 19 through 23 are acceptable models for non-LOCA transient analysis for CE80+ applications.[33] If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer initiates an evaluation.

The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:

a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

b. Conservative scram characteristics are assumed, i.e., for a PWR - maximum time delay with the most reactive rod held out of the core, and for a BWR - a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, doppler coefficient, axial power profile, and radial power distribution.

d. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. Compliance with Regulatory Guide 1.105 is determined by ~~ICSB~~HICB.[34]

Technical Rationale

The technical rationale for application of these acceptance criteria to reviewing events initiated by an increase in heat removal by the secondary system is discussed in the following paragraphs.[35]

(a) Compliance with GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 10 is applicable to this section because the reviewer evaluates the consequences of four anticipated operational occurrences that have the potential to exceed allowable thermal design criteria for fuel cladding integrity. These four anticipated operational occurrences involve the transient increase in heat removal by the secondary system, which in turn causes reactor power to increase in response to the resultant lowering of the temperature of the reactor coolant. Regulatory Guide 1.53 provides guidance with respect to the application of the single failure criterion to the design and analysis of nuclear power plant protection systems. Regulatory Guide 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for the four anticipated operational occurrences evaluated in this SRP section involving excessive heat removal by the secondary system.[36]

(b) Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC 15 is applicable to this section because the four overcooling events cause the reactor coolant system pressure to change in response to the drop in reactor coolant temperature. Although most of these events cause the reactor coolant pressure to decrease, some cause reactor coolant pressure to increase, depending on the worst single failure assumed. For example, for the ABWR the most severe initiating event in this group is a feedwater controller failure during maximum demand (runout of two feedwater pumps). This results in an increase in reactor pressure, but the increase is well within the ASME Code limit. Therefore, for the four overcooling transients of SRP Section 15.1.1, the reactor coolant pressure needs to be analyzed to ensure that the pressure acceptance criterion is satisfied.

Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the four anticipated operational occurrences evaluated in this SRP section involving excessive heat removal by the secondary system.[37]

(c) Compliance with GDC 20 requires that the reactor protection system be designed to initiate the operation of appropriate systems automatically, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences.

GDC 20 is applicable to this section because the reviewer evaluates the reactor protection system that operates to shut down the reactor automatically to terminate the events (anticipated operational occurrences) analyzed in this SRP section. The events are terminated by the reactor protection system in a timely manner such that fuel cladding integrity is maintained. For a BWR, this means that the minimum value of the critical power ratio reached during the transient should be such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. This limiting value of the minimum critical power ratio (MCPR) is called a safety limit. For the ABWR its value is 1.07. For a PWR, this means that the minimum value of the departure from nucleate boiling ratio (DNBR) reached during the transient must remain above the 95/95 DNBR limit for the applicable DNBR correlation. For the CE80+ design, this value is 1.24.

Meeting the requirements of GDC 20 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that the reactor protection system acts in a timely manner to terminate reactor operation prior to reaching a safety limit.[38]

(d) Compliance with GDC 26 requires that one of the reactivity control systems be control rods capable of reliably controlling reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

GDC 26 is applicable to this section because the reviewer evaluates four overcooling events analyzed in this section that may involve the movement of control rods in response to the initiating event, and rod misalignment, including stuck rods, can produce more severe thermal-hydraulic conditions than would otherwise exist. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. SRP Section 15.1.1 examines these margins where applicable to ensure that the thermal criteria limits are not exceeded.

Meeting the requirements of GDC 26 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that there is appropriate margin for malfunctions of the reactivity control system, including stuck rods.[39]

III. REVIEW PROCEDURES

The procedures below are used for both the construction permit (CP), ~~and~~ operating license (OL), and combined license (COL)[40] reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values are used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

~~RSB~~SRXB[41] reviews the applicant's description of the transients caused by excessive heat removal with specific attention to the occurrences that lead to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (per II.~~3.~~[42]b) are accounted for.

If the SAR states that a particular ~~transient~~ initiating event involving an increase in heat removal is not as limiting as some other similar ~~transient~~ event, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the increase-in-heat-removal event that is determined to be most limiting. For this ~~transient~~ event, the ~~RSB~~SRXB[43] reviewer, with the aid of the ~~ICSB~~HICB[44] reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to an acceptable level. The ~~RSB~~SRXB[45] reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ~~ICSB~~HICB[46] review of Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the ~~RSB~~SRXB[47] reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the SRP sections for Chapters 4, 5, 6, 7, 8, and 9 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by ~~RSB~~SRXB[48] to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the models is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by ~~RSB~~SRXB.[49] Of particular importance are the values of reactivity coefficients and control rod worths used by the applicant in this analysis, and the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates the justification provided by the applicant to show that the core burnup selected yields the minimum margins. ~~CPB~~SRXB ~~is consulted regarding~~reviews[50] the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared ~~to~~ with[51] the acceptance criteria presented in subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems. ~~The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system (if applicable) are reviewed.~~ Time-related variations of the following parameters are reviewed:

– reactor power;
– heat fluxes (average and maximum);
– reactor coolant system pressure;
– minimum DNBR (PWR) or CPR (BWR);
– core and recirculation loop coolant flow rates (BWR);
– coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions);
– steam line pressure;
– containment pressure;
– pressure relief valve flow rate; and
– flow rate from the reactor coolant system to the containment system (if applicable).[52]

The values of the more important of these parameters, as listed in subsection I of this SRP section, are compared ~~to~~ with those predicted for other similar plants to see that they are within the range expected.

The NRC has ~~undertaken~~completed[53] a program to reduce the sensitivity of B&W plants to feedwater transients, with emphasis on overcooling events that have occurred at B&W plants[54] (Items II.E.5.1 and II.E.5.2, NUREG-0660 and 0718). ~~When this program is complete, the RSB reviewer, with the aid of other branches as appropriate, should incorporate the program results into the review of this SRP section.~~This sensitivity is attributed to a number of design features including the small secondary water inventory in the once-through steam generators and a relatively small pressurizer.

Concerns regarding steam generator overcooling are related to the potential for loss of natural circulation due to bubble formation and the high frequency of high-pressure safety injection actuation during the transients. These transients may produce undesirable pressure/temperature conditions (pressurized thermal shock) that may cause excessive cycles of safety-related equipment, such as thermal cycling of safety injection nozzles and operation of primary system safety relief valves. A related concern is the possible overfilling of the steam generators by which water may be introduced in the steam lines, producing loads beyond the design basis.

The resolution of these concerns consists of design modifications that provide for automatic auxiliary feedwater flow and steam generator level control, and main feedwater overfill protection. The resolution of Items II.E.5.1 and II.E.5.2 is contained in References 25, 26, and 27.[55]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[56]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and ~~his~~ that the[57] review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

> A number of plant transients can result in an unplanned increase in heat removal by the secondary system. Those that might be expected to occur with moderate frequency can be caused by feedwater system or pressure regulator malfunctions or the inadvertent opening of a steam generator safety or relief valve (PWR only). All of these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _____ transient.
>
> The staff concludes that the analysis of transients resulting in an unplanned increase in heat removal by the secondary system that are expected to occur with moderate frequency is acceptable and meets the requirements of General Design Criteria 10, 15, 20,[58] and 26. ~~and TMI Action Plan items II.E.5.1 and II.E.5.2.~~[59]
>
> 1. In meeting ~~GDC~~ General Design Criteria 10, 15, 20,[60] and 26 as indicated below we have determined that the applicant's analysis was performed using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. In addition, we have further determined that the positions of Regulatory Guide 1.53 as related to the single failure criterion and Regulatory Guide 1.105 for instruments have also been satisfied.
>
> 2. The applicant has met the requirements of ~~GDC~~ General Design Criteria 10, 20,[61] and 26 with respect to demonstrating that resultant fuel ~~damage~~integrity[62] is maintained since the specified acceptable fuel design limits were not exceeded for this event.
>
> 3. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.
>
> 4. The applicant has met the requirements of ~~GDC~~ General Design Criteria 20 and[63] 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the specified acceptable fuel design limits were not exceeded.
>
> 5. The applicant has met the requirements of II.E.5.1 and II.E.5.2 by properly accounting for all design modifications in the analysis that has been made as a result of resolution of this item.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[64]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[65] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[66]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGS.

VI. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components." Article NB-7000, "Protection against Overpressure," American Society of Mechanical Engineers.

3. Standard Review Plan Section 4.2, "Fuel System Design."

4. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973.

6. "Reference Safety Analysis Report - RESAR-3,"Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973. "Reference Safety Analysis Report - RESAR-3S," Westinghouse Nuclear Energy Systems, July 1975; and "Reference Safety Analysis Report - RESAR-414," Westinghouse Nuclear Energy Systems, October 1976.

7. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

8. "Standard Nuclear Steam System B-SAR-205," Babcock & Wilcox Company, February 1974.

9. General Design Criterion 10, "Reactor Design."

10. General Design Criterion 15, "Reactor Coolant System Design."

11. General Design Criterion 20, "Protection System Functions."[67]

~~11~~12.[68] General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

~~12~~13. Regulatory Guide 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Systems Protection."

~~13~~14. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

~~14~~15. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident."

~~15~~16. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

17. General Electric Company, ODYNA - One Dimensional Dynamic Model (proprietary computer software for use in ABWR transient analysis to simulate pressurization events).[**] [69]

18. General Electric Company, REDYA (proprietary computer software for use in ABWR transient analysis to simulate other than pressurization events).[70]

19. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)[71]

20. TORC (CENPD-161) and CETOP (CENPD-206-P-A). (TORC is used to simulate the three-dimensional fluid conditions within the reactor core. Results from TORC include the core radial distribution of the relative channel axial flow that is used to calibrate CETOP. TORC or CETOP is used for DNBR calculations using the CE-1 critical heat flux correlation.)[72]

[**] The previously approved ODYN and REDY codes have been modified by GE for use in the analysis of limiting transients on the standard design Advanced Boiling Water Reactor (ABWR). These modified codes, ODYNA and REDYA, were reviewed by the NRC staff and have been approved for design analysis of the ABWR.

21. HERMITE (CENPD-188-A). (HERMITE is used to determine short-term response of the reactor core during the postulated reactor coolant pump rotor-seizure event and total loss-of-flow event.)[73]

22. COAST (SSAR; CENPD-98). (Calculates the time-dependent reactor coolant mass flow rate in each loop during reactor coolant pump coastdown transients.)[74]

23. STRIKIN-II (CENPD-133; CENPD-135 Supps. 2 and 4). (Calculates the cladding and fuel temperatures for an average or hot fuel rod.)[75]

24. 10 CFR 50.34(f)(2)(xvi), Additional TMI-Related Requirements, derived from TMI Action Plan Item II.E.5.1 contained in NUREG-0660 and NUREG-0718.[76]

25. NRC Memorandum dated March 15, 1983, Mattson to Dircks, "Closeout of NUREG-0660 Item II.E.5.1 Design Sensitivity of B&W Plants for Operating Plants."[77]

26. NRC Memorandum dated September 28, 1984, Denton to Dircks, "Closeout of TMI Action Plan Task II.E.5.2, Transient Response of B&W-Designed Reactors."[78]

27. NUREG-0793, Midland Safety Evaluation Report and Supplement 1, Section 5.5, "Design Sensitivity of B&W Reactors."[79]

28. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Report NEDO-24154 and NEDO-24154P, Volumes I, II, and III, October 1978.[80]

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SRP Draft Section 15.1.1

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
2.	Current secondary review branch name and designation	Added the current secondary review branch name and designation, PERB.
3.	Editorial	Changed "transients" to "events" or "initiating events," as appropriate, throughout this section. The transients discussed in this section are frequently initiating events that can lead to pressure and temperature transients.
4.	Editorial	Defined "SRP" as "Standard Review Plan."
5.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
6.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
7.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
8.	SRP-UDP format item	Added Review Interfaces subheading under Areas of Review.
9.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
10.	SRP-UDP format item	Broke the one large paragraph describing review interfaces into lettered paragraphs, one for each review interface. The existing order and text was preserved, except for necessary updating for branch names and designations and where noted.
11.	Current interfacing review branch	Changed the current interfacing review branch name and designation, HICB.
12.	Integrated Impact No. 783	Added a sentence to the interface review responsibilities of HICB to indicate that it is responsible for reviewing the applicant's design criterion for the allowable number of actuation cycles of the emergency core cooling system and the reactor protection system.
13.	Current interfacing review	Changed the current interfacing review for the review of SRP Sections 4.2 through 4.4, SRXB.
14.	Current secondary review branch	Changed the current secondary review branch name and designation, PERB.
15.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
16.	Current interfacing review branch	Changed the current interfacing review branch name and designation, TSB.
17.	Editorial	Deleted superfluous words "as being reviewed" to improve clarity.
18.	Editorial	Added the word "primary" for consistency.
19.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
20.	Editorial	Provided "GDC 10" as initialism for "General Design Criterion 10."
21.	Editorial	Changed "assure" to "ensure" (global change for this section).
22.	Editorial	Provided "GDC 15" as initialism for "General Design Criterion 15."
23.	Integrated Impact No. 1351	Added GDC 20 to Acceptance Criteria for AOOs.
24.	Editorial	Provided "GDC 26" as initialism for "General Design Criterion 26."
25.	Editorial	Relettered paragraphs due to insertion of new paragraph C above.
26.	Integrated Impact No. 783	Deleted paragraph D under ACCEPTANCE CRITERIA. It is believed that this paragraph was placed here as a flag to alert the reviewer to the possibility of changes to the transients in this SRP section for B&W plants. These plant modifications that resulted from these TMI Action Plan items are now known and are discussed under REVIEW PROCEDURES for general information and because these transients may need to be recorded if emergency core cooling system actuation or reactor protection system actuation occurs. 10 CFR 50.34(f)(2)(xvi) requires applicants to establish a design criterion for the allowable number of such actuation cycles consistent with the expected occurrence rates of severe overcooling events, considering both anticipated transients and accidents (applicable to B&W designs only). These requirements were derived from TMI Action Plan Item II.E.5.1.
27.	Editorial	Changed "GDC" to "General Design Criteria" (global change for this section).
28.	Integrated Impact No. 1351	Added GDC 20 to Acceptance Criteria for AOOs.
29.	Editorial	Added a comma after "error," for correct usage.
30.	Integrated Impact No. 1351	Added GDC 20 to Acceptance Criteria for AOOs.

Item	Source	Description
31.	SRP-UDP format item	Deleted "(Ref. 12)" in accordance with standard practice to delete redundant references.
32.	Integrated Impact No. 784	Added a sentence referring to References 17, 18 and 28. These are the ODYN, ODYNA and REDYA computer codes approved by NRC for transient analysis of BWRs (ODYN) and the ABWR (ODYNA and REDYA).
33.	Integrated Impact No. 785	Added an sentence referring to References 19 through 23. These are ABB-CE topical reports approved by NRC for non-LOCA transient and accident analysis of CE80+ plants.
34.	Current interfacing review branch	Changed the current interfacing review branch designation, HICB.
35.	SRP-UDP format item	Added Technical Rationale subheading and introductory paragraph under Acceptance Criteria.
36.	SRP-UDP format item	Added technical rationale related to GDC 10.
37.	SRP-UDP format item	Added technical rationale related to GDC 15.
38.	SRP-UDP format item	Added technical rationale related to GDC 20.
39.	SRP-UDP format item	Added technical rationale related to GDC 26.
40.	SRP-UDP format item	Added a reference to combined license (COL) reviews.
41.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
42.	Editorial	Changed the correct referenced paragraph to II.b.
43.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
44.	Current interfacing review branch	Changed the current interfacing review branch designation, HICB.
45.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
46.	Current interfacing review branch	Changed the current interfacing review branch designation, HICB.
47.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
48.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.
49.	Current primary review branch designation	Changed the current primary review branch designation, SRXB.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
50.	Current primary review branch designation	The old interfacing review branch was deleted and SRXB substituted because SRXB is now responsible for this review.
51.	Editorial	Changed "compared to" to "compared with" (global change for this section).
52.	Editorial	Revised a complex sentence to improve clarity.
53.	Integrated Impact No. 783	Revised to indicate that the program to reduce the sensitivity of B&W plants to feedwater transients has been completed.
54.	Integrated Impact No. 783	Added a phrase to the sentence on overcooling to better characterize the nature of the study. This phase was taken from the closeout letter for NUREG-0660 Item II.E.5.1 (reference 25).
55.	Integrated Impact No. 783	Added a discussion of the resolution of Items II.E 5.1 and II.E.5.2 of NUREG-0660 and NUREG-0718. The text was taken from the closeout memos for these items (references 25 and 26 for SRP Section 15.1.1).
56.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
57.	Editorial	Modified to eliminate gender-specific reference.
58.	Integrated Impact No. 1351	Added GDC 20 to the Evaluation Findings.
59.	Integrated Impact No. 783	Same as note 20 above.
60.	Integrated Impact No. 1351	Added GDC 20 to the Evaluation Findings.
61.	Integrated Impact No. 1351	Added GDC 20 to the Evaluation Findings.
62.	Editorial	Deleted the word "damage," and substituted the word "integrity," in order for the sentence to convey the intended meaning.
63.	Integrated Impact No. 1351	Added GDC 20 to the Evaluation Findings.
64.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
65.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
66.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.

Item	Source	Description
67.	Integrated Impact No. 1351	Added GDC 20 to the list of references as an acceptance criterion for AOOs.
68.	Editorial	Since Reference 11 has been added, References 11 through 15 have been renumbered as shown.
69.	Integrated Impact No. 784	Added the ODYNA computer code as Reference 17.
70.	Integrated Impact No. 784	Added the REDYA computer code as Reference 18.
71.	Integrated Impact No. 785	Added the CESEC-III computer code as Reference 19.
72.	Integrated Impact No. 785	Added the TORC and CETOP computer codes as Reference 20.
73.	Integrated Impact No. 785	Added the HERMITE computer code as Reference 21.
74.	Integrated Impact No. 785	Added the COAST computer code as Reference 22.
75.	Integrated Impact No. 785	Added the STRIKIN-II computer code as Reference 23.
76.	Integrated Impact No. 783	Added a reference to 10 CFR 50.34(f)(2)(xvi).
77.	Integrated Impact No. 783	Added the closeout reference for Item II.E.5.1 of NUREG-0660.
78.	Integrated Impact No. 783	Added the closeout reference for Item II.E.5.2 of NUREG-0660.
79.	Integrated Impact No. 783	Added a reference to the "lead plant" safety evaluation related to the design sensitivity of B&W reactors to feedwater transients (the Midland SER and Supplement 1).
80.	**Integrated Impact 784**	Added reference to the ODYN computer code (GE topical report NEDO-24154). The association of ODYN with GE topical report NEDO-24154 is established in the ABWR FSER Section 15.1. The title for NEDO-24154 was verified from references in the ABWR FSER and SRP Section 15.2.1, Reference 4.

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Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
781	Consider updating Regulatory Guide 1.105 to the current version of ISA-S67.04.	No changes were made to SRP Section 15.1.1 as a result of this Integrated Impact.
782	Consider updating Regulatory Guide 1.53 to the current version of IEEE Std 379.	No changes were made to SRP Section 15.1.1 as a result of this Integrated Impact.
783	Incorporate the resolution of TMI Action Plan Items II.E.5.1 and II.E.5.2 into SRP Section 15.1.1.	I.A AREAS OF REVIEW II. ACCEPTANCE CRITERIA (old acceptance criteria D was deleted) III. REVIEW PROCEDURES (3 places) IV. EVALUATION FINDINGS VI. REFERENCES. References 24, 25, 26, and 27
784	Add the ODYNA and REDYA computer codes to the ACCEPTANCE CRITERIA and REFERENCES for ABWR transient analysis.	II. ACCEPTANCE CRITERIA VI. REFERENCES. References 17 and 18
785	Add the ABB-CE topical reports (computer codes) for non-LOCA transient analysis of the CE80+ plant to the ACCEPTANCE CRITERIA and REFERENCES.	II. ACCEPTANCE CRITERIA (paragraph following paragraph 6) VI. REFERENCES. References 19 through 23
1351	Add GDC 20 to ACCEPTANCE CRITERIA, EVALUATION FINDINGS, and REFERENCES.	II.C, ACCEPTANCE CRITERIA II., ACCEPTANCE CRITERIA (introductory paragraph to paragraphs 1 through 6 II.5, ACCEPTANCE CRITERIA II.(c), Technical Rationale IV., EVALUATION FINDINGS (third paragraph) IV.1, EVALUATION FINDINGS IV.2, EVALUATION FINDINGS IV.4, EVALUATION FINDINGS VI., REFERENCES, Reference 11